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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

A	projected area of the human body in the neutron direction
cm	centimeter
DE_n	first collision dose equivalent from neutrons
DE_γ	first collision dose equivalent from gamma rays
D_n	first collision absorbed dose from neutrons
$D_n(H)$	first collision absorbed dose from neutrons to a human
DOE	U.S. Department of Energy
D_γ	first collision absorbed dose from gamma rays
$D_\gamma(H)$	first collision absorbed dose from gamma rays to a human
eV	electron volt
ft	feet
g	gram
gal	gallon
hr	hour
ICRP	International Commission on Radiological Protection
in.	inch
IREP	Interactive RadioEpidemiological Program (a computer program)
keV	thousand electron volts
kg	kilogram
L	liter
MeV	million electron volts
min	minute
mL	milliliter
mrem	millirem
n	neutron
NIOSH	National Institute for Occupational Safety and Health
ORAU	Oak Ridge Associated Universities
POC	probability of causation
r	distance from center of reactive solution in barrel to the beltline of an individual exposed during the accident
RBE	relative biological effectiveness
RPG	radiation protection guideline
s	second
S	specific activity

$S_{wb}(B)$ activity of ^{24}Na per mL of whole blood from burro
$S_{wb}(H)$ activity of ^{24}Na per mL of whole blood from human

TIB technical information bulletin

U.S.C. United States Code
UCCND Union Carbide Corporation – Nuclear Division

V volume of the human body

wk week
w_R radiation weighting factor from ICRP Publication 60

yr year

Φ neutron fluence

μCi microcurie

§ section

$\bar{\xi}$ spectrum-average capture probability for neutrons within the human body

1.0 INTRODUCTION

Technical information bulletins (TIBs) are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained. TIBs may be used to assist the NIOSH staff in the completion of individual work required for each dose reconstruction.

In this document, the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)” [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work.

As noted above, the statute includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition contains an exclusion with respect to the Naval Nuclear Propulsion Program, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external dosimetry monitoring results are considered valid for use in dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived:

[1] The U.S. Department of Labor is ultimately responsible under the EEOICPA for determining the POC.

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

2.0 PURPOSE

The purpose of this TIB is to review the available dosimetric data and its potential application in dose reconstruction for Y-12 workers who were near the nuclear criticality accident in Building 9212 of the Y-12 Plant in Oak Ridge, Tennessee, in 1958. The accident occurred on Monday, June 16, at approximately 2:05 p.m. (UCNC 1958; Callihan and Thomas 1959). Additional data relevant to individuals described in this TIB of a personal nature can be found in *Y-12 1958 Criticality Accident Roster* (ORAUT 2006a), which is available to dose reconstructors as needed.

3.0 IDENTIFICATION AND INTERROGATION OF EXPOSED INDIVIDUALS

By 2:45 p.m. on Monday, June 16, 1958, radiation surveys of workers for evidence of personal contamination and for indications of neutron activation of indium foils in security badges were underway at two control centers. Starting in 1955, strips of indium foil with a mass of approximately 1 g had been included in the Y-12 security badges for all employees (McLendon 1959). Indium has a very large capture cross section for neutrons, and the beta and gamma radiation from the ^{116m}In isotope (half-life of 56.3 min) produced by neutron activation of the stable ^{115}In isotope are easily measured. The measured activity of the foils is not very useful in radiation dosimetry, but is useful as a quick technique for sorting nuclear accident exposures in order of most to least severe. This quick search procedure was helpful in (1) preventing unnecessary overloading of medical facilities by lightly exposed workers, (2) providing reassurance to workers who had received small or negligible exposures, and (3) providing an immediate indication of the seriousness of the nuclear criticality accident to management.

Interrogation of the workers assembled at the control centers was also begun in an attempt to establish the exact location of the accident. Very high readings were detected from the indium foil in the security badge of an employee (Employee A) who worked as a chemical operator in C-1 Wing. A process supervisor questioned the employee and concluded that the accident must have occurred in C-1 Wing of Building 9212. It was eventually determined that eight employees were in close proximity to the accident. The above chemical operator was within 3 to 6 ft, and the other seven were at distances ranging from 15 to 50 ft. A more complete assessment of the indium foils in the security badges of all employees indicated a total of 31 individuals with potentially significant neutron doses from the accident (UCNC 1958). Table 3-1 of this TIB provides a summary of certain available data for these 31 individuals, and ORAUT (2006a) contains additional personal data on the employees that dose reconstructors can access as needed. The identification numbers and letters for individuals in Table 3-1 and ORAUT (2006a) are taken directly from UCNC (1958), Y-12 Plant (1958) and Hurst, Ritchie, and Emerson (1959).

Only five individuals in the vicinity of the 1958 criticality accident were wearing a film badge dosimeter (see Table 3-1). The Y-12 policy at the time was to issue film badge dosimeters to workers if their potential dose from external radiation exposure exceeded 10% of established radiation protection guidelines (RPGs). The RPGs in 1958 were 1.5 rem/wk (75 rem/yr) to the extremities, 600 mrem/wk (30 rem/yr) to the skin, and 300 mrem/wk (15 mrem/yr) to the whole body (ORAUT 2005a,b). Gamma radiation from highly enriched uranium is normally not the controlling challenge to radiation protection (DOE 2004). However, the interaction of alpha particles from uranium with nuclei of fluorine and other low-atomic number atoms generates neutrons with energies of approximately 2 MeV. The magnitude of the neutron flux varies, based on the total activity of uranium (a function of enrichment) and the chemical compound in question (ORAUT 2005c). In general, an individual's exposure to neutrons

Table 3-1. Information on individuals near the accident.

Employee identification		Job title	External monitoring (years)	2nd quarter 1958 gamma/beta dose (mrem)	Distance from criticality (ft)
Number	Letter				
1	A	Chemical Operator	1961–81		3–6
2	B	Electrician	1961–88		14
3	C	Maintenance Mechanic	1961–72		17
4	D	Electrician	1961–72		22
5	E	Maintenance Mechanic	1961–80		22
6	F	Welder	1961–76		25
7	G	Maintenance Mechanic	1961–64		25
8		Equipment Tabulation Clerk	1954–57, 61–67		29
9	H	Chemical Operator	1958, 61–65	47/34[a]	50
10		Chemical Operator	1961–62		71
11		Chemical Operator	1961–85		85
12		Chemical Operator	1961–78		92
13		Chemical Operator	1961–78		100
14		Health Physics Inspector	1951–53, 61–80		137
15		Electrician	None		148
16		Process Foreman	1950 (1 qtr.), 61–82		175
17		Chemical Operator	1955, 57–62	333/247	178
18		Draftsman Engineer	1952, 61–87		183
19		Engineer	1961–87		190
20		Chemical Operator	1953–59, 61–80	333/250	212
21		Chemical Operator	1961–86		216
22		Record Clerk	1961–68		224
23		Chemist	1952–54, 61–84		260
24		Stenographer	1961–73		276
25		Chemist Associate	1953–55, 58, 61–88	167/432	280
26		Laboratory Analyst	1961–85		288
27		Laboratory Analyst	1952–54, 61–82		304
28		Development Engineer	1961–72		312
29		Product Inspector	1953–72	29/51	340
30		Road Foreman	1961–87		390
31		Receiving Clerk	1958–62	291/521	412

a. On supplemental film badge program from Feburary 1 to April 15, 1958. Not wearing a film badge at time of accident.

generated by (alpha,neutron) reactions was low unless the individual was required to spend more than a few hours per week in close proximity to large storage containers of uranium fluoride compounds or spending time near storage or processing areas for large amounts of uranium fluoride compounds (DOE 2004). Therefore, most individuals working in chemical process areas for enriched uranium at the Y-12 Plant were only monitored on a routine basis for internal uranium exposures, and the contaminants in recycled enriched uranium were controlled at the Y-12 Plant so that both internal and external exposures were characteristic of those from uncontaminated enriched uranium (BWXT Y-12 2000).

Following the accident, a new film badge was developed for use at all DOE facilities operated by the Union Carbide Corporation – Nuclear Division (UCCND). This film badge served as a security badge and also provided for monitoring of both routine and accident-related radiation exposures (Thornton, Davis, and Gupton 1961; Hurst and Ritchie 1961; McRee, West, and McLendon 1965). Starting in January 1961, all employees at the Y-12 Plant and at other DOE facilities operated by UCCND were monitored using this combination security badge and radiation dosimeter.

4.0 DESCRIPTION OF BUILDING 9212 COMPLEX AND ACCIDENT

The 9212 Building complex at the Y-12 Plant includes Buildings 9212, 9809, 9812, 9815, 9980, and 9981 (ORAUT 2005d; BWXT Y-12 2003; UCNC 1958). Buildings 9980 (Radiography) and 9981 (X-Ray) were an integral part of Building 9212 as shown in Figure 4-1 (UCNC 1958). The accident occurred in an area in which salvable enriched uranium (approximately 90% ^{235}U) was recovered from various materials by physicochemical methods (Callihan and Thomas 1959).

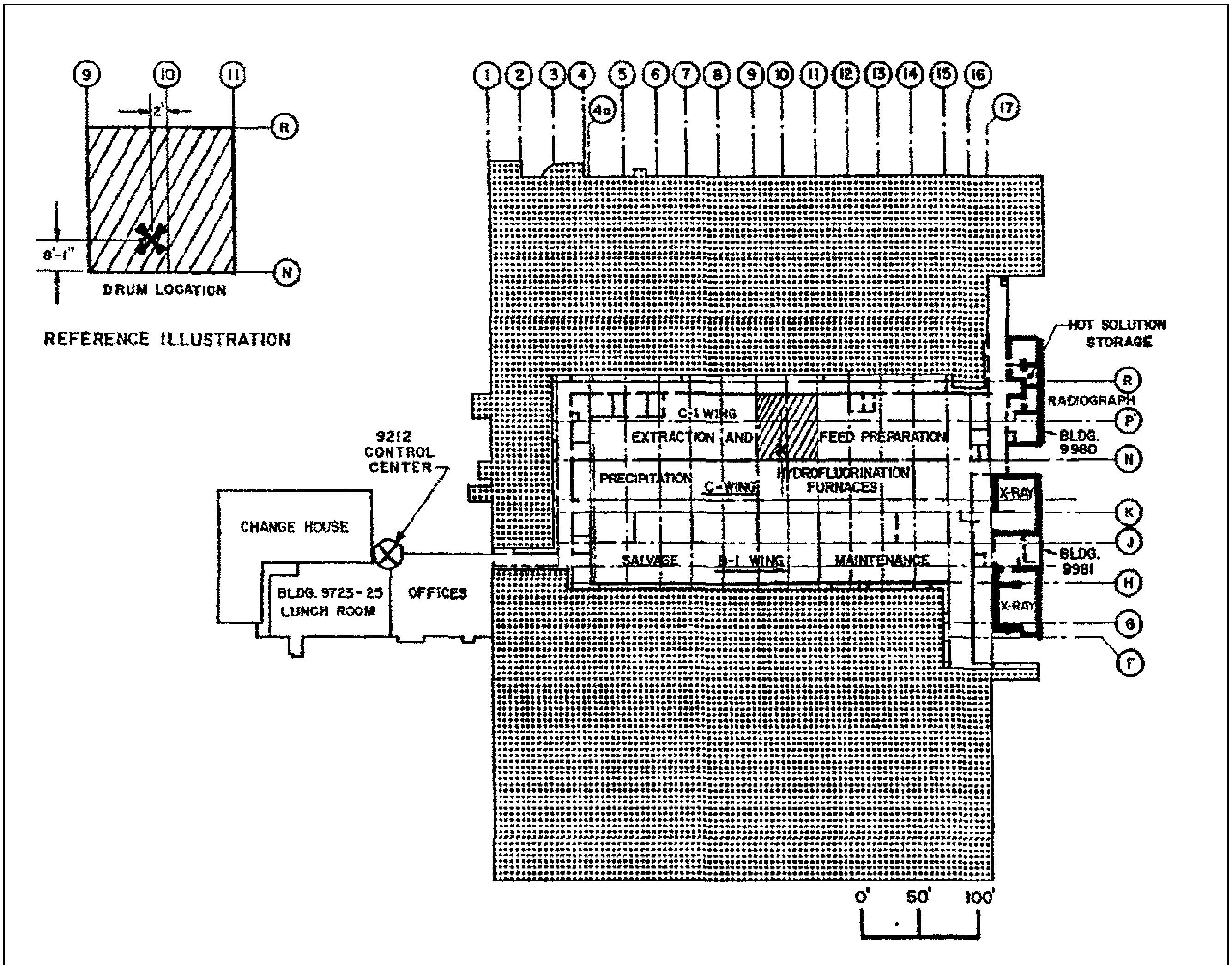

Figure 4-1. Building 9212 with location of accident.

Over 100 different processes could be performed in the Building 9212 complex. Building 9212, the largest building, was constructed in the early 1940s. It was a multistory steel frame structure infilled at the perimeter with hollow clay tile. The original mission of Building 9212 was to recover highly enriched uranium from the electromagnetic separation process. The original structure consisted of a central building (the Headhouse) 72 ft wide by 308 ft long (N-S direction) and four parallel wings projecting from the east side of the Headhouse, each 36 ft by 254 ft (the A, B, C, and D Wings). The open space between the wings was designed to help mitigate the effects of a nuclear criticality accident or a chemical explosion. In the late 1940s, Building 9212 was expanded to accommodate the increased production of uranium from the Oak Ridge Gaseous Diffusion Plant and to recover highly enriched uranium from scrap materials. In the late 1940s, new structures were erected in the spaces between the existing A, B, and C Wings (these were called the A-1, B-1, and C-1 Wings) and an adjoining D-Wing (the D-1 Wing). A single-story steel frame structure 133 ft wide and 400 ft long was added in the early 1950s (the E-wing) adjacent to the D-1 Wing north of the Headhouse. The E-Wing was used to cast and machine enriched uranium components.

The process phase in which the accident occurred was a temporary arrangement involving portions of a new installation in the startup stage (B-1 Wing) and an old installation in the shutdown stage (C-1 and C Wings). This situation arose from delays in the startup of the new facilities in B-1 Wing for the conversion of uranium nitrate solution to uranium tetrafluoride. At the time of the accident, the uranium processing areas in Building 9212 were concerned with the required monthly accounting of uranium in inventory, which necessitated a stoppage of operations. The method of taking inventory varied with the form and concentration of the uranium. In certain process areas, where equipment contained dilute homogenous uranium solutions, a satisfactory accounting could be made by taking samples and computing the contents of known volumes. In the process area where the accident occurred, more precise accounting was required because of the higher uranium concentrations and the tendency of the solutions to deposit uranium-bearing solids in the equipment. This more precise accounting was obtained by processing the contents of several safe-geometry tanks for uranium tetrafluoride just before inventory. In addition, the process was to wash, dismantle, and swab out the safe-geometry tanks, collecting the washings in portable safe-geometry bottles. After reassembly, the safe-geometry tanks were prone to leak at the tank ends. Therefore, the tanks were filled with water, checked for leaks, and the water drained from the tanks before their return to operation.

In the interval between reassembly and leak testing, a highly enriched uranium nitrate solution had accumulated in the tanks through a valve that was supposed to provide isolation from operating equipment in the B-1 and C-1 Wings. The water being drained from the tanks was preceded by this solution. The accident occurred when a critical volume of uranium nitrate solution drained into a 55-gal drum during this operation in which only water was expected to flow from the safe-geometry tanks of C-1 Wing. The drum was about 33 in. (84 cm) in height and 22 in. (56 cm) in diameter. A volume of 56.2 L containing 2.1 kg of ^{235}U and standing at a height of 23.5 cm was estimated as the delayed critical configuration.

There were no strong ambient neutron fields in C-1 Wing. Therefore, the system might have been prompt critical before any fissions occurred in the drum, and the neutron initiating the first fission reaction in the drum might have come from an (alpha,neutron) reaction in the uranium nitrate solution. Once fission started, the power level rose rapidly to a high level, and the fission energy produced gas bubbles by dissociation of the solution, which reduced its density and drove the system subcritical. Escape of the gas bubbles allowed the system to return to prompt critical, and with the delayed neutrons as a source, the power level rose again. This cycling probably continued for several minutes as the temperature of the solution increased. A high solution temperature would cause the system to settle into delayed critical and subsequently into a subcritical configuration due to the continued flow of water into the drum from the safe-geometry tanks in C-1 Wing. It was estimated that about 1×10^{17} fissions occurred during the first pulse and 1×10^{18} fissions occurred over a period of about 20 min. There was no evidence that solution in the drum splashed out of the drum during the accident or caused serious contamination to the area surrounding the accident (McLendon 1959).

5.0 EXTERNAL DOSE ESTIMATES FOR THE EIGHT MOST HIGHLY EXPOSED INDIVIDUALS

Very early estimates of the radiation doses received by the eight most highly exposed workers, based on their known locations at the time of the accident and an estimate of the number of fissions during the accident, were unreasonably high (Hurst, Ritchie, and Emerson 1959). For example, the fast neutron component of the total radiation dose received by Employee A was 1,500 rad. It is assumed, therefore, that the location of the eight most highly exposed employees and their exposure geometries were unknown during a significant part of their exposure. These facts, together with the fact that the eight most highly exposed employees were not wearing personnel monitoring devices, dictated that sodium activation in the body be used in the evaluation of their radiation doses (Hurst, Ritchie, and Emerson 1959).

The capture of thermal neutrons by ^{23}Na in the body gives rise to ^{24}Na, which has a half-life of 15 hr and emits a 1.37-MeV gamma ray in cascade with a 2.76-MeV gamma ray per ^{24}Na decay (Shleien, Slaback, and Birky 1998; Martin 2000). The human body is several mean free paths thick for fast neutrons. As a consequence and as shown in Figure 5-1, the probability that a fast neutron will be captured within the body as a thermal neutron is not a very sensitive function of the initial energy of the fast neutron (Cross 1981). The specific activity *S* of ^{24}Na in the body is related to the capture probability $\bar{\xi}$ as follows:

$$S = 7.5\,\bar{\xi}\,\frac{A}{V}\,\Phi \cong 0.61\,\bar{\xi}\,\Phi \text{ µBq } {}^{24}\text{Na/g } {}^{23}\text{Na} \qquad \text{Eq. 5-1}$$

where *V* is the volume of the body, *A* is the projected area of the body in the neutron direction, Φ is the total neutron fluence at all energies, and $\bar{\xi}$ is the spectrum-average capture probability for neutrons within the body (IAEA 1982). The capture probability data shown in Figure 5-1 are for

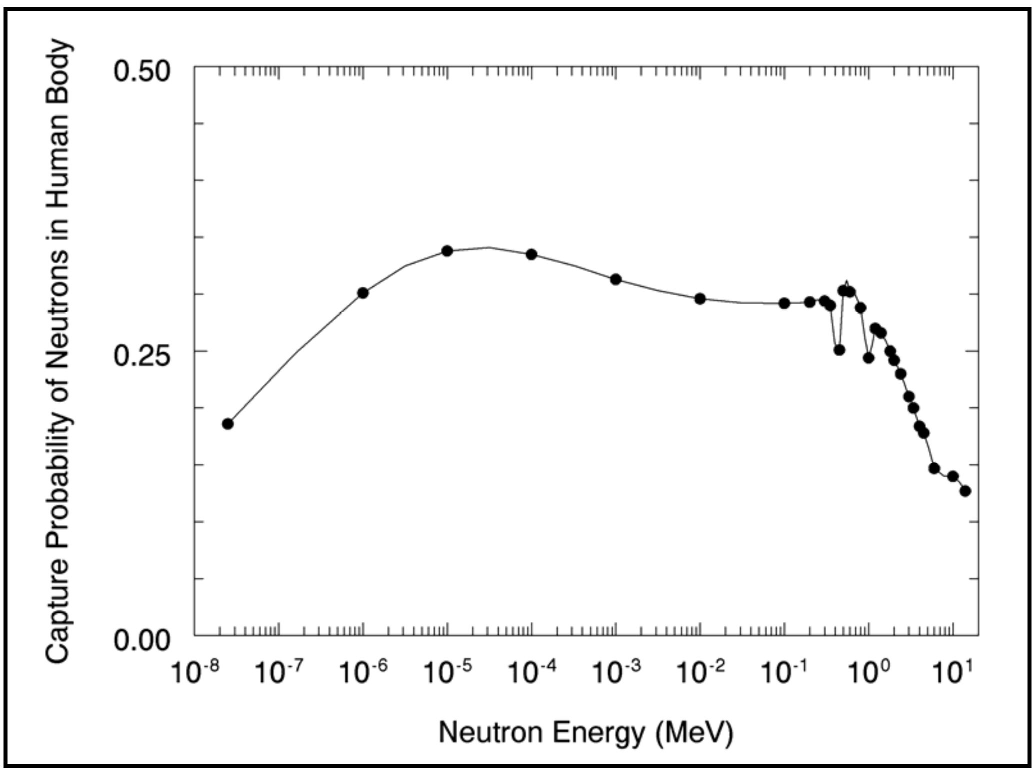

Figure 5-1. Thermal neutron capture probability as a function of the energy of neutrons normally incident on the front or back of the human body (Cross 1981).

neutrons normally incident on the front or back of the body (Cross 1981). For neutrons normally incident on the sides of the body, both *A* and *S* are approximately 20% less. Therefore, there is an uncertainty of as much as 20% in an estimate of the neutron dose component if an individual's body orientation in the neutron field is not known or if the orientation changes during the accident.

The neutron and gamma-ray doses to the eight individuals whose indium foils indicated the highest exposures were determined by the use of blood sodium activation. At approximately 5:00 pm on the day of the accident, 100 mL of blood was collected from several of the most highly exposed individuals (Hurst, Ritchie, and Emerson 1959; UCNC 1958). Each blood sample was placed in a small polyethylene bottle and the ^{24}Na activity was determined using a scintillation counter. During the morning of the next day, a second set of blood samples was collected from all of the highly exposed individuals. This time only 50-mL blood samples were collected, an anticoagulant (heparin) was added to prevent clotting, and the blood samples were counted as before. The second set of measurements was used as the basis for dose estimation. To establish the relationship between neutron dose and blood sodium activation, a mock-up of the criticality accident was constructed and operated as a low-power reactor in two experiments (UCNC 1958; Hurst, Ritchie, and Emerson 1959). During the first experiment, the first collision doses from neutrons and gamma-rays were measured at a distance of about 200 cm (6 ft). The neutron dose was measured with an absolute fast-neutron dosimeter (Hurst and Ritchie 1961), and the gamma-ray dose was measured with an ionization chamber having carbon walls and CO_2 gas (Hurst and Ritchie 1962; Ballweg and Meem 1951). After making a correction for the difference in the fission product gamma rays from the reactor mockup, a value of 2.8 was estimated as the ratio of the first collision doses from gamma rays and neutrons for the accident (Hurst, Ritchie, and Emerson 1959). During the second experiment, a burro was exposed at the same distance from the mockup to a first collision dose of 48 rad from fast neutrons. The burro was chosen because its torso was comparable in size to that of a man and the amount of sodium per gram of blood serum is nearly the same for burro and man (Hurst, Ritchie, and Emerson 1959; Auxier, Sanders, and Snyder 1961). Blood samples were collected from the burro and counted for ^{24}Na activity in the same manner as the second set of measurements discussed above. The blood sodium activation in the burro was determined to be 2.9×10^{-4} µCi/mL of whole blood from a first collision dose of 48 rad from fast neutrons (Hurst, Ritchie, and Emerson 1959). Table 5-1 provides a summary of the first collision doses from fast neutrons and gamma rays for the mostly highly exposed individuals (i.e., Employees A to H in Table 3-1). The last column of Table 5-1 lists the doses of record for these eight individuals (UCNC 1958).

The first collision absorbed doses from fast neutrons and gamma rays in Table 5-1 were calculated using the following relationships:

$$D_n(H) = \frac{S_{wb}(H)}{S_{wb}(B)} \times 48 \text{ rads, and } D_\gamma(H) = 2.8 \times D_n(H) \qquad \text{Eq. 5-2}$$

where $S_{wb}(B)$ and $S_{wb}(H)$ are the activities of ^{24}Na per mL of whole blood from the burro and human, respectively; 48 rad is the first collision neutron dose to the burro; $D_n(H)$ and $D_\gamma(H)$ are the first collision doses to the human from neutrons and gamma rays, respectively; and 2.8 is the gamma-to-neutron dose ratio for the exposures. A previous review of the dosimetry for the accident suggested that the estimated first collision doses based on blood sodium activation should be increased by approximately 10% (Mole 1984). Because more than 90% of the blood sodium is contained in the blood serum, the standard practice today is to use the sodium activation in blood serum rather than the sodium activation in whole blood (Kerr and Mei 1993). The haematocrit, or the proportion of a

Table 5-1. Doses of record for the eight most highly exposed individuals.[a]

Exposed individuals	Blood sodium activation (µCi/mL)	First collision adsorbed dose (rad)			First collision dose equivalent (rem)
		Neutrons	Gamma rays	Total	
A	5.8E-4	96	269	365	461
B	4.3E-4	71	199	270	341
C	5.4E-4	89	250	339	428
D	5.2E-4	86	241	327	413
E	3.7E-4	62	174	236	298
F	1.1E-4	18	50.5	68.5	86.5
G	1.2E-4	18	50.5	68.5	86.5
H	3.6E-5	6.0	16.8	22.8	28.8
Burro	2.9E-4	48	N/A[b]	N/A	N/A

a. See UCNC (1958). The first collision dose equivalent assumes a relative biological effectiveness (RBE) of 2 for the neutron component of first collision absorbed dose based on deterministic effects related to lethality (Langham 1967; ICRP 1990). The estimated uncertainty in both the neutron and gamma-ray components of absorbed dose for these eight individuals is 20% (Hurst, Ritchie, and Emerson 1959).
b. N/A = not applicable.

blood sample by volume that consists of red blood cells, was 40% to 47% for the eight most highly exposed workers and 36.5% for the burro (Brucer 1958). Therefore, the amount of blood serum in the sample from the burro was larger by about 10% than the amount of serum in the human samples, and the recorded dose estimates for all exposed individuals at the accident have been increased by 10% in this report as recommend in the 1984 review by Mole (see Table 5-2).

Table 5-2. Correction to recorded doses for the eight most highly exposed individuals.[a]

Exposed individuals	First collision neutron dose (rad)		First collision gamma-ray dose (rad)	
	UCNC (1958)	Mole (1984)	UCNC (1958)	Mole (1984)
A	96	106	269	296
B	71	78	199	219
C	89	98	250	275
D	86	95	241	265
E	62	68	174	191
F	18	20	50.5	55.6
G	18	20	50.5	55.6
H	6.0	6.6	16.8	18.5

a. The use of sodium activation in whole blood rather than the current standard use of sodium activation in blood serum caused the recorded neutron and gamma-ray doses for the eight most highly exposed individuals at the Y-12 1958 criticality accident (UCNC 1958) to be underestimated by approximately 10% (Mole 1984).

6.0 EXTERNAL DOSE ESTIMATES FOR OTHER INDIVIDUALS

The distances in Table 3-1 were used to estimate the first collision absorbed doses to other individuals near the accident. These are the distances from the center of the reactant solution in the barrel to individuals' beltlines at the time of first criticality (UCNC 1958). Therefore, the radiation doses are based on the neutron dose from fissions occurring during the first pulse and gamma-ray dose from the prompt fission gamma rays, neutron capture gamma rays, and fission product gamma rays emitted during the first 15 s after the first pulse. This is consistent with the estimation of radiation doses in the previous section for the eight most highly exposed individuals (UCNC 1958; Hurst, Ritchie, and Emerson 1959). The 15 s represents the time taken for most individuals to exit from Building 9212 after the evacuation alarms sounded. The radiation doses for the two individuals who were 25 ft from the accident are well established from blood sodium activation (i.e., Employees F and

G), so their doses were used to estimate radiation doses to other individuals at larger distances by the application of $1/r^2$ scaling. The relationships used in this scaling are:

$$D_n = [25 \text{ ft}/r]^2 \times 20 \text{ rad, and } D_\gamma = [25 \text{ ft}/r]^2 \times 55.6 \text{ rad} \qquad \text{Eq. 6-1}$$

where D_n, D_γ, and r are the first collision neutron dose, first collision gamma-ray dose, and exposure distance in Table 3-1, respectively, for each individual of interest. Table 6-1 summarizes the estimated neutron and gamma-ray doses for other individuals near the accident, and Figure 6-1 shows the estimated gamma-ray doses in comparison to the available film badge data for five of the employees (see Tables 3-1 and 6-1).

Table 6-1. Estimated doses for other individuals near the accident.

Exposed individual	Distance from criticality (ft)	First collision neutron dose (rad)	First collision gamma-ray dose (rad)
8	29	14.9	41.3
10	71	2.48	6.89
11	85	1.73	4.81
12	92	1.48	4.11
13	100	1.25	3.48
14	137	0.666	1.85
15	148	0.571	1.59
16	175	0.408	1.13
17	178	0.395	1.10
18	183	0.373	1.04
19	190	0.346	0.963
20	212	0.278	0.773
21	216	0.268	0.745
22	224	0.249	0.693
23	260	0.185	0.514
24	276	0.164	0.456
25	280	0.159	0.443
26	288	0.151	0.419
27	304	0.135	0.376
28	312	0.128	0.357
29	340	0.108	0.301
30	390	0.0822	0.228
31	412	0.0736	0.205

The buildup and attenuation of the in-air neutron and gamma-ray fields should be offsetting factors at the distances of interest here, but there could have been significant shielding of some individuals near the accident due to the equipment in the building and the building's walls. For example, four of the gamma-ray doses for individuals wearing film badges showed a potential for significant shielding against radiation from the accident (i.e., Employees 17, 20, 25, and 29), while the film badge data for Employee 31 was in reasonably close agreement with the estimated gamma-ray doses (Figure 6-1). For dose reconstruction purposes, the estimated neutron and gamma-ray doses for Employee 31 in Table 6-1 should be used as a default value for other individuals who (1) were in or near Building 9212 at the time of the accident but (2) whose names and social security numbers are not among the 31 names and social security numbers listed in ORAUT (2006a).

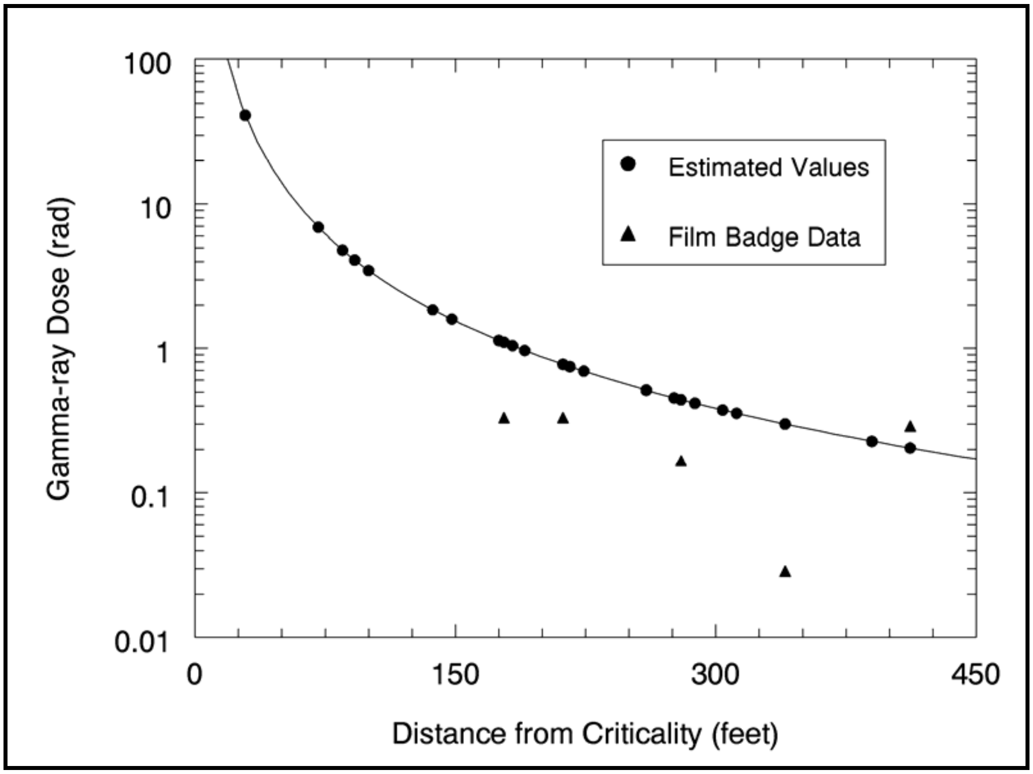

Figure 6-1. Comparison of estimated and measured gamma-ray doses for other individuals near the accident (see Tables 3-1 and 6-1).

7.0 EXTERNAL DOSE MODIFICATION FOR USE IN IREP

When historical radiation measurements are used in dose reconstruction, the factors used to account for the greater relative biological effectiveness of neutrons compared to gamma rays must be removed such that absorbed dose is the fundamental unit and the radiation weighting factors w_R from the International Commission on Radiological Protection's Publication 60 (ICRP 1991) must be applied before the conversion to organ dose for calculation of the probability of causation of a specific cancer using the Interactive RadioEpidemiological Program (IREP) (NIOSH 2002; ORAUT 2006b). The emphasis in this TIB has been on the estimation of absorbed doses to individuals near the accident. Therefore, it is only necessary to apply the weighting factors to the absorbed doses in Tables 5-2 and 6-1 of this report.

IREP uses three gamma-ray energy ranges and five neutron energy ranges to calculate the organ dose of interest. The three gamma-ray energy ranges are less than 30 keV, 30 to 250 keV, and more than 250 keV; the five neutron energy ranges are less than 10 keV, 10 to 100 keV, 0.1 to 2 MeV, 2 to 20 MeV, and more than 20 MeV (NIOSH 2002), which are the same as the ones used by ICRP to define the w_R values for neutrons (ICRP 1991). Table 7-1 lists the calculated neutron fluence and absorbed dose distributions published in Table K.11 of Y-12 Plant (1958). Note that distribution data from the report are given in terms of six neutron energy ranges from thermal to 10 MeV. As a result, the estimated percent of absorbed dose for the IREP neutron intervals was calculated by regrouping the energy internals in Table 7-1.

Table 7-1. Theoretical neutron energy distribution (UCNC 1958).

Neutron energy range	Percent total neutrons, Φ (n/cm^2)	Percent absorbed dose, D_n (rad)
Thermal (0–0.04 eV)	35.8	13.1
0.04 eV–5 keV	9.2	0.4
5–750 keV	14.0	8.0
0.75–1.5 MeV	8.4	11.5
1.5–2.5 MeV	10.9	18.0
2.5–10 MeV	21.8	49.0

For the first IREP neutron energy interval (less than 10 keV), all of the first two energy intervals in Table 7-1 were combined and the total percentage of the absorbed dose at energies greater than 10 keV was 13.5%. Because the second IREP neutron energy group (10 to 100 keV) is relatively small in comparison to the 5- to 750-keV neutron energy range in Table 7-1, the absorbed dose was assumed to be evenly distributed across this larger neutron energy group or approximately 1% per 100 keV. This corresponds to a 1% contribution to the absorbed dose for the second IREP neutron energy group of 10 to 100 keV. The remaining 7% of the absorbed dose from the 5- to 750-keV interval was combined with the 11.5% from the 0.75- to 1.5-MeV energy range and 9% (or half) of the 1.5- to 2.5-MeV energy range for a total of 27.5% in the third IREP neutron energy group of 0.1 to 2 MeV. The remaining 9% in the 1.5- to 2.5-MeV energy range was then combined with the 49% from the 2.5- to 10-MeV energy range for a total of 58% in the fourth IREP neutron energy group from 2 to 20 MeV. There was no absorbed dose specified for the fifth IREP neutron energy group more than 20 MeV.

Table 7-2 lists the sample dose equivalents for each IREP neutron energy range calculated from the neutron absorbed dose of 6.6 rad for Employee H (9) (see Tables 3-1and 5-2), the ICRP Publication 60 (ICRP 1991) radiation weighting factors, and the dose fractions for the neutron energy ranges. The calculated total neutron dose equivalent of 79.7 rem in Table 7-2 is significantly greater than the recorded neutron dose equivalent of 12 rem for Employee H. However, the recorded neutron dose equivalent of 12 rem is based on a neutron RBE of 2 for deterministic effects related to lethality (Langham 1967; ICRP 1990), while the calculated neutron dose equivalent in Table 7-2 is based on the Publication 60 weighting factors for stochastic effects such as cancer that are of concern at the low doses and dose rates normally encountered in radiation protection (ICRP 1991, 2003). The Publication 60 radiation weighting factors yield dose equivalent estimates for both neutrons and gamma rays that are extremely claimant-favorable at the large dose rates and doses involved in the accident (i.e., doses from neutrons were delivered in a few seconds and the doses from gamma rays were delivered over about 15 s).

Table 7-2. Sample calculation of the neutron dose equivalent for Employee H (9).

IREP neutron energy range	ICRP 60 weighting factor, w_R	Dose fraction (%)	Estimated absorbed dose, D_n (rad)	Estimated dose equivalent, DE_n (rem)
<10 keV	5	13.5	6.6	4.46
10–100 keV	10	1.0	6.6	0.66
0.1–2 MeV	20	27.5	6.6	36.30
2–20 MeV	10	58.0	6.6	38.28
>20 MeV	5	0	6.6	0
Total				79.7

Table 7-3 lists the dose equivalent estimates for both neutrons and gamma rays for reconstruction of organ doses for the cancer of interest. The neutron dose equivalents are broken down into the neutron energy groups used in IREP, and the gamma-ray dose equivalent is based on a radiation weighting factor of 1 for gamma rays of all energies (ICRP 1991). For individuals located at 50 feet or less from the criticality (see Table 3-1), the gamma rays are lumped together in the one IREP energy group of more than 250 keV because of the very high energies of the prompt gamma rays from fission, the capture gamma rays from fission neutron interactions, and the delayed gamma rays from the highly radioactive fission products (IAEA 1982). At larger distances, it is expected that there would be scattered gamma rays in the 30- to 250-keV energy range due to shielding of an individual by walls and other equipment in the building. Thus, 25% of the gamma rays at distances of more than 50 feet from the criticality are assumed to be in the 30- to 250-keV energy group and 75% in the more than 250 keV energy group. Because of the way the IREP computer program treats gamma rays with energies of 30-250 keV, this would be more favorable to the claimant than the assumption of 100% gamma rays with energies of more than 250 keV. As recommended previously, the dose equivalent estimates for Employee 31 should be used as claimant favorable dose estimates for other employees who were in or near Building 9212 at the time of the criticality accident but are not among the 31 individuals listed in ORAUT (2006a).

Table 7-3. Dose equivalent estimates for all individuals near the accident.

Exposed individual	Neutron dose equivalent, DE_n (rem)					Gamma dose equivalent, DE_γ (rem)
	<10 keV	10–100 keV	0.1–2 MeV	2–20 MeV	Total	
1	71.6	10.6	583.0	614.8	1,280	296
2	52.7	7.8	429.0	452.4	942	219
3	66.2	9.8	539.0	568.4	1,183	275
4	64.1	9.5	522.5	551.0	1,147	265
5	45.9	6.8	374.0	394.4	821	191
6	13.5	2.0	110.0	116.0	242	55.6
7	13.5	2.0	110.0	116.0	242	55.6
8	10.1	1.5	82.0	86.4	180	41.3
9	4.46	0.66	36.30	38.28	79.7	18.5
10	1.67	0.25	13.64	14.38	29.9	6.89
11	1.17	0.17	9.52	10.03	20.9	4.81
12	1.00	0.15	8.14	8.58	17.8	4.11
13	0.84	0.12	6.88	7.25	15.1	3.48
14	0.450	0.067	3.663	3.863	8.04	1.85
15	0.385	0.057	3.141	3.312	6.89	1.59
16	0.275	0.041	2.244	2.366	4.93	1.13
17	0.267	0.040	2.173	2.291	4.77	1.10
18	0.252	0.037	2.052	2.163	4.5	1.04
19	0.234	0.035	1.903	2.007	4.18	0.963
20	0.188	0.028	1.529	1.612	3.36	0.773
21	0.181	0.027	1.474	1.554	3.24	0.745
22	0.168	0.025	1.370	1.444	3.01	0.693
23	0.125	0.019	1.018	1.073	2.23	0.514
24	0.111	0.016	0.902	0.951	1.98	0.456
25	0.107	0.016	0.875	0.922	1.92	0.443
26	0.102	0.015	0.831	0.876	1.82	0.419
27	0.091	0.014	0.743	0.783	1.63	0.376
28	0.086	0.013	0.704	0.742	1.55	0.357
29	0.073	0.011	0.594	0.626	1.30	0.301
30	0.0555	0.0082	0.4521	0.4768	0.993	0.228
31	0.0497	0.0074	0.4048	0.4269	0.889	0.205

8.0 SUMMARY

This TIB provides an important update to the radiation dosimetry for the accident (UCNC 1958; Hurst, Ritchie, and Emerson 1959). The dose estimates for the eight most highly exposed individuals were based on measurements of their blood sodium activation and a mockup of the criticality using a burro as a surrogate for man. Measurements of sodium activation in blood from the burro made it possible to estimate the first collision absorbed dose from neutron and gamma rays for these eight workers. An RBE of 2 based on deterministic effects related to lethality was then used to estimate the first collision dose equivalent for the eight workers. The uncertainty of the first collision doses was estimated to be approximately 20% (Hurst, Ritchie, and Emerson 1959).

A 1984 review of the radiation dosimetry for the accident suggested that the estimated first collision doses based on blood sodium activation should be increased by approximately 10% (Mole 1984). Because more than 90% of the blood sodium is contained in the blood serum, the standard practice today is to use the sodium activation in blood serum rather than sodium activation in whole blood (Kerr and Mei 1993). The haematocrit or the proportion of a blood sample by volume that consists of red blood cells was 40% to 47% for the eight most highly exposed workers and 36.5% for the burro (Brucer 1958). Therefore, the amount of blood serum in the sample from the burro was larger by about 10% than the amount of serum in the human blood samples, and the recorded doses for all eight of the most highly exposed workers were corrected to account for the 10% increase recommended in the 1984 review by Mole (1984). The estimated uncertainties of 10% by Mole and 20% by Hurst, Ritchie, and Emerson (1958) suggest an overall uncertainty of approximately 25% in the estimated dose to workers from the accident.

Information on distances of workers from the accident at the time of exposure was available for another 23 workers (Y-12 Plant 1958). The first collision doses from neutrons and gamma rays for these workers were estimated using $1/r^2$ scaling of the radiation fields and normalized to the first collision doses for two of the eight highly exposed workers who were 25 ft from the accident. The dose estimates were then compared with available film badge data for 5 of the 23 workers. Four of the gamma-ray doses for workers wearing film badges showed a significant potential for shielding against radiation from the accident, while the film badge data for the individual at the greatest distance from the criticality was found to be in reasonably close agreement with the estimated gamma-ray dose. Therefore, the estimated gamma-ray and neutron doses for this individual should be used as the default dose values for any other workers in or near Building 9212 at the time of the accident.

To obtain the first collision dose equivalent as required by the IREP computer program (NIOSH 2002), ICRP Publication 60 radiation weighting factors were applied (ICRP 1991). The dose equivalent estimates for the eight most highly exposed workers are quite large in comparison to the recorded dose equivalents in the literature (UCNC 1958; Hurst, Ritchie, and Emerson 1959) because the recorded dose equivalents are based on an RBE of 2 for deterministic effects in relation to lethality (Langham 1967; ICRP 1990), while the dose equivalent estimates calculated here are based on radiation weighting factors for stochastic effects such as cancer that are of concern at the low doses and dose rates normally encountered in routine radiation protection (ICRP 1991, 2003). The Publication 60 radiation weighting factors yield dose equivalent estimates for both neutrons and gamma rays (Table 7-3) that are extremely claimant-favorable at the large dose rates and doses involved in the accident.

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